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Abstract

CONTINUOUS METHODS OF TRACING CHEMICAL CHANGES IN FLUIDS UNDER GAMMA IRRADIATION[x]

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Two methods: spectrophotometry and polarography have been adopted for the work in strong fields /of the order 100 rads per sec/ of gamma radiation. All factors influencing the measurement have been investigated. In the result of continuous measurement, kinetic curves are obtained showing changes of concentrations of different involved species in the function of the time of irradiation and the time after the irradiation. Sensitivities of the methods are 10^{-7} to 10^{-5} M/l, depending on the substance. Therefore at the dose rate of 100 rads/sec and the radiation yield G=1, the kinetic curves may be traced during the irradiations lasting only few seconds. The sensitivity may be reduced by several orders of magnitude so that kinetic curves of the G value equal even to 10000 /found in our laboratory/ may be followed. High sensitivity of the measurement enables us to obtain kinetics of medium-stable intermediates of radiolysis, namely /in the case of water/, of hydrogen peroxide and indirectly of peroxyorganic compounds, occuring as intermediates in chain oxydation reaction, initiated by radiation.

The developed techniques of continuous analysis during the irradiation served in research performed on several systems. Among them were systems enabling the classification of organic compounds according to their reactivity in irradiated solutions. Our methods made it possible to investigate the radiation chemistry of neutral /pH = 5-8,5/ solutions of ferrous salts. Methods of direct analysis during irradiation enabled us to find and investigate the chain reactions in alkaline solutions as well as a closer approach to the radiation chemistry of the neglected field of high pH-values.

With our methods a study was performed on industrial application of radiation induced oxidation of iron. Positive results were reached in cases, where organic compounds may be present in the solution.

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x/ Under contract no 43 with the International Atomic Energy Agency

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Abstract

RADIATION CHEMISTRY IN ALKALINE SOLUTIONS

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The oxyanions of manganese proved to be a particularly convenient object of investigations of the radiolysis of the alkaline solutions, because the hydrolyses complicating the reaction do not occure here. The reduction of permanganate under gamma irradiation /^{60}Co about 200 c activity/ proceeds successively: $MnO_4^- \longrightarrow MnO_4^{2-} \longrightarrow MnO_4^{3-} \longrightarrow MnO_2$. Each subsequent stage occurs after the preceding one has been finally accomplished. Thus it was possible to investigate seperately each of these stages by selecting the suitable concentration of OH^- ions. The radiation yield of each particular stage in alkaline solutions may be expressed as below: $G(-MnO_4^{n-}) = G_H + 2G_{H_2O_2} - G_{OH} = 1.9$ equivalents per 100 eV. This yield is much lower than in neutral solutions, which is due to the fact that OH radicals occur here in their proper role of oxidizing agents. They oxidize all manganese oxyanions in which manganese is on a lower than +7 oxidation state. In neutral solutions OH radicals participate directly in the reduction of permanganate, thus causing the yield increase in comparison with alkaline solutions. They cause also the yield dependence on the initial concentration of permanganate. This yield is expressed by an empirical expression: $G(-MnO_4^-) = 1.7 + 1.62 \cdot 10^2 MnO_4^{-\,1/2}$, which has been also theoretically derived.

This twofold role of OH radicals e.g. as oxidizing and reducing agents, has been confirmed also in the radiolysis of potassium iodide solution. OH radicals oxidize I^- to I_2 and reduce the hypoiodite formed in neutral or alkaline solutions with I_2. Thus, OH radicals induce the considerable increase of the radiation yield with the increase of the initial I^- concentration, while causing its decrease with the increase of pH.

In alkaline solutions the radiation yield of the reduction of manganese oxyanions does not depend on the initial concentration of oxyanions nor on the concentration and type of hydroxide. Hence the general conclusions have been drawn concerning the model of radiolysis of the alkaline solutions in which the radicals are formed directly from OH^- and Me^+ ions. This model has been confirmed also in the radiolysis of the chromate and chromite solutions. In these solutions the chromites are oxidized to chromates by gamma rays. Chromates however, are reduced to chromites only in solutions saturated by hydrogen with a yield which depends on the concentration ratio CrO_2^- to H_2.

On the basis of the proposed mechanism of the radiative reduction of manganate in concentrated alkaline solutions the following molecular yields in these conditions have been calculated: $G_{H_2} = 0.12$; $G_{H_2O_2} = 0.51$

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Abstract

PIPE-LINES WEAR IN IRON ORE TRANSPORTATION

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A study on the wear of steel pipes carrying fines of iron ore in water suspension regarding the viability of a pipe-line with about 350 km is emphasized. So far 20 experiments were conducted during one year of research. The method consisted in the insertion of radioactivated tubes (about 70 cm length) in experimental loops consisting of 1" diameter tubes and 20 m length. For this radioactivation, a small reactor (TRIGA MARK I) has been used and specific activities of about 5 μci/g of Fe-59 were obtained. Each experiment was conducted for about 200 hours. Samples were counted in well-scintillators of NaI(Tl); the countings were translated in worn mass by reference to standards of known mass irradiated together with the tubes. The experiments correlated the effect of the granulometry on the life of the tubes; also the slurry concentration, transport velocity and grinding techniques were considered in the total wear. The study also deals with the separated effects of the abrasion and corrosion. The contribution of corrosion seems to be small. A decreased wear rate was found in the progress of each experiment due to the smoothening of the grains caused by collisions. Photomicrographs were made to check the shape of the grains. A significant design decrease of the sacrifice thickness of the pipe-line walls in the direction of the flow is expected, should economical and hydraulical reasons imply high wear rates. One typical result gives a wear of 0.2 mm/year of eroded thickness at the beginning of the steel pipe-line (C - 0.22%; Mn - 0.48%; P - 0.019%; S - 0.013%; Si - 0.27%) if the slurry circulates with 2.2 m/sec carrying 50% in weight of hematite ground to 90% less than 325 mesh and 10% between 200 and 325 mesh.

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Abstract

RESEARCH REACTOR PROGRESS IN PAKISTAN

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For training in reactor operation and maintenance and for research using reactors, Pakistan has so far relied heavily on facilities abroad. About 50 engineers and more than 100 scientists have had extensive training at laboratories in USA, UK, Canada and other advanced countries. Since 1961 training courses of one year duration have been conducted at the Atomic Energy Centre, Lahore, in reactor technology. A natural uranium water-moderated sub-critical assembly has been set up at Lahore and is used in conjunction with a neutron generator for steady state and time dependent reactor studies. Staff members are also engaged in research work on various problems in reactor physics and engineering.

Pakistan's first research reactor is of the pool type with a power level of 5 MW. It is under construction as part of the Pakistan Institute of Nuclear Science and Technology (PINSTECH) at Nilore, Islamabad, 14 miles east of Rawalpindi. Phase-I of the Institute which has the research reactor as its central facility with five attached laboratories is to be completed by June 1965. Reactor criticality is scheduled for June 1965. Following a period of low-power tests the reactor will be available for research work at full power towards the end of 1965.

Unlike some other countries Pakistan has delayed the installation of a large research reactor until a corps of experienced scientists and engineers was available who could utilize the reactor facilities fully. When the research reactor is installed at PINSTECH in 1965 there will therefore be a sufficient number of technical persons to assure efficient utilization of the reactor. Emphasis is currently being given to the manufacture of neutron diffraction equipment and other research apparatus. A reactor oscillator is also being constructed. Arrangements for the manufacture of long- and short-lived radioisotopes for use in Pakistan are being made. Projects in radiation biology and radiation chemistry are also being finalised.

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Abstract

Operation and Experimental Utilization of the High-Flux Research Reactor DR 3

by

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The reactor DR 3, belonging to the Danish Atomic Energy Commission, is a replica of the English Pluto reactor: 10 MW thermal power, D_2O moderated and cooled, with 26 fuel elements of the MTR type and a maximum thermal flux in the middle of the core of approx. 1.6×10^{14} n/cm^2 sec. It was built by the English company Head Wrightson Processes, Ltd., in 1957-59.

Criticality was reached in January 1960 and full power in October. Since April 1961 the reactor has been operated at full power in periods of 4 weeks with an overall on-power time of approx. 75%.

Reactor operation and experiments have not been attended with any serious trouble. 230 fuel elements have all worked excellently. 6 containers with spent fuel (150 elements) have successfully been sent to Dounreay in Scotland for reprocessing. The D_2O concentration, originally 99.79% by weight, has dropped at a rate of 0.002%/month to 99.70%. The pD has been lowered from 6.2 to 5.6 to diminish the rate of corrosion. The water is still clear without any visible sign of Al-hydroxide. The control system has worked satisfactorily. Leakage problems in the fine-control rod will be finally solved by substituting a D_2O-cooled rod. The safety rods will be replaced by more effective ones.

During full power operation 92 unwanted shut-downs have occurred, 52% of which were caused by failure of the mains. No radiation damage to persons has occurred, and evacuation of staff from the reactor building has never been necessary.

Kinetic experiments have been made to elucidate some of the problems of failure of primary cooling and to provide a basis for a reactor model. This

model is not yet capable of handling powers above approx. 20 MW, but will be modified in the near future to cover higher power ranges.

Occasionally, many power dips of 200-2000 kW, lasting for approx. 10-15 sec, have been observed. Rise of power has occurred very few times. Investigations have been concentrated on the possibility of mechanical vibrations of components in the core causing this trouble. Measurements have shown that the noise spectrum varies appreciably, sometimes containing heavy components of frequencies around 1 Hz. Extensive measurements have been planned for the spring of 1964.

The experiments may be divided into two categories, beam experiments with high-quality neutron beams and materials testing experiments, some of which have been carried out for or in co-operation with foreign countries and for the Dragon project. Graphite, uranium, steel, and zircalloy have been irradiated.

Among planned experiments may be mentioned an organic-loop irradiation, for the reactor study project DOR, of a fuel rod consisting of SAP and UO_2, with OM 2 as coolant. The working pressure will be 24 at and T_{out} 425° C for investigation of fouling, heat transmission and metallurgical problems.

Another experiment is being planned to elucidate the mechanical properties of SAP and UO_2 pellets and the interaction between these materials under in-pile conditions.

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Abstract

RADIOCHEMICAL INVESTIGATIONS AND CHEMICAL RESEARCH ASSOCIATED TO NUCLEAR REACTORS IN BRAZIL

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A report on radiochemical investigations and chemical research associated to nuclear reactors in three major Brazilian centres is presented.

A brief survey of chemical work associated to radioactivity carried out in Brazil prior to the installation of research reactors in this country is reviewed.

Radiochemical investigations started as a result of development of Nuclear Physics with the installation of several machines such as a 30 Mev betatron, 4 Mev Van de Graaf and 1 Mev Cockroft-Walton. The installation of nuclear reactors sited at S. Paulo (5 Mw swimming-pool) and Belo Horizonte (Mark I Triga Reactor) entailed new possibilities for research.

Radiochemical research in the Centro Brasileiro de Pesquisas Físicas has covered and is progressing along the following lines: investigation on the production of radioelements in the atmosphere by cosmic ray induced nuclear reactions, physico-chemical investigations with radioisotopes such as complex formation in solutions of lanthanides and actinides, the electrochemistry of tracers, and, more recently, studies on molecular structure by the Mossbauer Effect.

The installation of the 5 Mw swimming-pool reactor in the Instituto de Energia Atômica, S. Paulo, lead to a series of works connected, at first, with the construction and assembling of the machine. For about two years, work was carried out along service-function lines such as maintenance and correlated

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problems on the general areas requiring chemical competence. Later on, with the construction of appropriate laboratory facilities, work developed along the main line of production of nuclear grade fissile material, using tracer techniques for the development of analytical methods, and of radioisotope production in research reactor.

At the same time research work was conducted, mainly, in the analytical radiochemistry field with the development of methods which had to be used for the analysis of nuclear grade materials such as uranium, graphite, beryllium, rare-earths elements, as well as for minerals concerning their contents for gold, uranium, niobium, tantalum and elements present in low concentration.

Activation analysis and isotope dilution analytical procedures were worked out and applied for the solution of the above purely chemical problems.

On a latter stage, efforts were devoted to the analytical steps of application of radioisotopes to technology, hydrology and chemical fabrication of organic compounds.

Research is now being conducted on short-lived radioisotopes by association of instrumental techniques and fast radiochemical separations such as focusing ion exchanger and focusing electrophoresis.

The addition of a Triga Reactor to the Instituto de Pesquisas Radioativas, Belo Horizonte, together with radiation chemistry facilities allowed for the development of new radiochemical-research lines in that Centre. Prior to the reactor installation, research on chemical problems was carried out mainly on the analysis of minerals and on fall-out. Present research is being conducted in radiation chemistry with emphasis on the development of high dose dosimeters (neutron and gamma), and generally on the chemical consequences on nuclear transformations in solids and liquids. Tracer applications to some industrial problems such as the transport of minerals in pipe lines and on the extractive metallurgy of niobium and beryllium are being conducted. Activation analysis of samples of geological and industrial materials is now being made on a routine basis.

A survey of the Brazilian bibliography on radiochemistry is presented. The current research work is discussed together with a presentation of existing facilities in the country.

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Abstract

OBJECTIVES, ORGANIZATIONAL ASPECTS AND UTILIZATION OF THE IEA-R1 RESEARCH REACTOR.

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A brief description of the IEA-R1 research reactor, including irradiation and beam facilities, as well as some comments on the objectives of its utilization, are presented. Some items related to the overall organization such as: general management, operation section, safety committee, emergency procedure, health physics, and other supporting services, are discussed.

The scientific and technological work so far performed with the reactor, and the means available to accomplish this work, are treated in some detail. Emphasis is given to the collection of experimental data leading to better operational performance and utilization.

Experiments carried out with the reactor for education and training purposes are briefly described. Some of these experiments have been included in the curricula of courses on nuclear science and engineering, sponsored by the Brazilian Nuclear Energy Commission, and given by various institutions.

Radioisotope production, one of the main activities related to the use of the reactor, is discussed, so far as the irradiation, chemical processing, labelling of compounds, and distribution aspects are concerned.

Research work in nuclear physics is concentrated mainly in the field of nuclear data (neutron cross section measurements), and results so far obtained are summarized. The design of a neutron-gamma converter, to be installed in one of the reactor tangential channels, is under way. The use of this facility either for photonuclear or neutron capture gamma-rays studies are discussed. A solid state physics project is also mentioned. Neutron monochromators such as a crystal spectrometer and a mechanical velocity selector, designed and constructed at the IEA, were utilized in several projects. These instruments, a proton recoil chamber and other techniques were also used to specify the reactor neutron spectra in a broad range of energies.

Radiochemistry research covers activation and isotope dilution analysis and general tracer work with short and medium half-life radionuclides produced in the IEA-R1. Early work on activation analysis using single-channel pulse-heigh analysers and recent improvements , including the use of multichannel analysers and a special pneumatic rabbit station for short-lived radioisotopes, are mentioned.

The irradiation of biological material such as seeds,plants and small animals is considered, and sample preparation, irradiation devices and dosimetry, are discussed. The irradiation of equipment and some materials for special purposes is also described.

Some aspects of the cooperation with other scientific and technical institutions and future trends with regard to the reactor utilization, are presented.

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Abstract

SIX YEARS OF UTILIZATION OF THE POLISH RESEARCH REACTOR "EWA"

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The Polish Research Reactor "EWA" at Świerk /2MW, enriched uranium, moderator and coolant- light water, an average thermal neutron flux 10^{13} neutrons/sq.cm.sec/ became critical in 1958. Since that more than 300 papers and reports concerned with work directly connected with the reactor has been published.

The 9 horizontal experimental channels and the thermal column are permanently used for research in the field of solid state, nuclear and reactor physics. 3 triple axes neutron crystal spectrometers, a cold neutron facility /a beryllium filter and a slow chopper with curved slits/, 3 slow choppers, a fast chopper, a special apparatus for in-pile irradiation, a pile oscillator and some other additional equipment - all home made - enable research on crystal and magnetic neutron structure analysis, neutron-phonon and neutron-magnon interactions, microdynamics of molecules, fast neutron induced damage in semiconductors, neutron resonance scattering, /n,γ/ reactions /including short lived isotopes/, tri-partition of U 235, neutron diffusion, and neutron flux and spectra measurements.

Irradiation facilities within the reactor enabled activation analysis, chemistry of hot atoms and also the production of radioisotopes widely used in the country in the physical and chemical sciences, life sciences and industry.

The reactor "EWA" itself is also the object of extensive studies, changes and improvements /e.g. mechanical and ion-exchange filters for the first cooling circuit, corrosion studies, automatic control of slow reactivity changes etc./.

It undergoes a steady evolution with the aim to meet the actual and steady varying needs of the users. The power of the reactor will be increased to 4 MW in 1964 and plans are made to reach 10 MW by 1966. Using the same fuel elements as for "EWA" a very flexible and inexpensive zero power reactor "MARYLA" has been put into operation in the main reactor hall. This has facilitated to shift several experiments needing small power /e.g. the reactor oscillator/ from "EWA" to "MARYLA" and run the "EWA" reactor round the clock on full power.

Special ways and means to meet the various objectives of the reactor have been developed and established. The Laboratory for Reactor Exploitation has been created and it fulfils the task of reactor operation and development, trains specialists in this field and - on a limited scale - also in reactor physics and reactor design. In order to co-ordinate the fulfilment of the growing demand for experimental channels and irradiation space and sometimes contradictory needs of users a special reactor co-ordination and planning committe has been formed. This committee collaborates with the users of the reactor: Laboratories of the Institute of Nuclear Research, Institute of Nuclear Physics and a number of other scientific, technical and industrial institutions in the country. As a result of close contacts with universities many theses for MSc and PhD degress have been performed at the Reactor. Collaboration with other reactor centers abroad and with the Joint Institute of Nuclear Research in Dubna /USRR/ takes place, too.

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Abstract

RESEARCH REACTORS IN JAPAN

Atomic Energy Society of Japan

It was in the middle of the 1950's that the peaceful uses of atomic energy really started in Japan. In the summer of 1957, a small research reactor, the first reactor in Japan, went critical, and in the seven ensuing years to date, 2 large and 4 small research reactors as well as a training reactor have been completed. In addition, 3 other research reactors are at present under construction.

These reactors can be classified into three groups: 4 reactors located in one atomic center, the Japan Atomic Energy Research Institute (JAERI); 3 reactors designed and constructed by private companies for their own use; 4 reactors owned by universities. These reactors have been utilized in various ways to best suit the purpose of their installation: The reactors in JAERI, have been utilized in a wide range of uses from basic to applied or development research. The reactors owned by private companies have contributed to raising the technological level of the companies through the experience gained in manufacturing the equipment and components related to atomic energy and by their utilization of the reactors for developing the equipment and components. The reactors owned by universities have been utilized for education and training, as well as for basic and applied research in various fields.

Thus, although many research reactors have been constructed

in Japan in a relatively short period, the reactors have been effectively utilized, and are playing a major role in stimulating progress in the field by their use in training scientists and promoting the advance of technological competence in producing nuclear equipment.

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Abstract

THE DRAGON PROJECT

by

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The Dragon Project is a joint undertaking between twelve European countries, under the sponsorship of the European Nuclear Energy Agency of the Organisation for Economic Cooperation and Development, to carry out a programme of work in the field of graphite moderated, helium cooled, high temperature reactors. The history of the Project is first described, together with the more important provisions of the original Agreement of March 1959 for work in the five years from April 1959 and of the revised Agreement of November 1962 covering the eight-year period up to March 1967.

The Project's tasks, and the organisation and policies adopted for the execution of those tasks, are summarised, and a report of the present state of the Project's work is given.

The international aspects of the Project are examined and their effect upon the working of the Project is discussed. The relative advantages and disadvantages of the Project's form of organisation are reviewed.

A summary is given of the aims and objects of the programme of research and development work carried out in the Project. This work was initially directed to providing answers to specific problems which arose during the design phase of the reactor experiment but has been extended, wherever possible, to cover the more general problems arising in high temperature reactors. One very important aspect of the Project's research and development work has been, and is, the study of fuel elements for these reactors.

A description is given of the reactor experiment being built at Winfrith with particular emphasis on the features of this reactor experiment which should help in solving the problems of future high temperature reactors.

Finally, an account is given of the work which the Project has done on large high temperature reactors. The assumptions on which the work is based are set out and the steps being taken to prove or disprove these assumptions are discussed.

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Abstract

REVIEW OF RESEARCH AND DEVELOPMENT WORK FOR THE DRAGON PROJECT

by

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The uranium-thorium cycle in a large power reactor with carbon moderator is described briefly. In the Dragon reactor experiment, the core is small and it has been found desirable to employ two types of fuel arranged in two zones. One type consists of uranium-235, thorium-232 and carbon, in proportions that might be typical of the composition in a large power reactor: The other type of fuel contains uranium-235 and zirconium as monocarbides, and makes up about three quarters of the core. The main reactor physics features of this core arrangement are described.

The adoption of the purged fuel element in the Dragon reactor experiment has necessitated the development of special graphites.

The problems of obtaining the right physical characteristics in these graphites and the production difficulties are reviewed briefly. Although helium is used as a coolant, the presence of oxidising impurities leads to corrosion and mass transfer of the exposed graphite surfaces. The investigation of this phenomenon has been carried out in laboratory-scale experiments and also with out-of-pile and in-pile loops. A complementary programme of work on the purification of helium to remove the chemical impurities and entrained fission products has also been carried out on the laboratory scale and in a pilot plant.

The diffusion of fission products and their evolution from graphite/carbide fuels at high temperatures is one of the major problems in reactors of this type. A solution to the problem has been sought in the use of pyrocarbon and silicon carbide coated particles. Results of irradiation experiments with a variety of specimens of uranium/zirconium monocarbide and uranium/thorium dicarbide coated particle fuels are reviewed.

The production of coated particle fuel and its assembly into the Dragon reactor experiment fuel element is being carried out by the Project on the reactor site. An outline of the production procedure is given.

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Abstract

EXPERIENCE OF OPERATION AND USE OF THE U.K. HIGH FLUX RESEARCH REACTORS

contributed by staff of the U.K.A.E.A.
at Harwell, Dounreay & Aldermaston
edited by R. F. Jackson*

The principal features of the three similar U.K. heavy water high flux research reactors, DIDO and PLUTO at Harwell and the Dounreay Materials Testing Reactor are briefly recalled. The total space available is listed for experiments in the three main categories of radiation damage, isotope production and beam use. The allocation of this space is quoted for each of several main types of experiment, for example non-fissile damage studies. The breakdown of effort in aid of specific power projects, general reactor technology, basic research and isotope production is also given.

Brief reference is made to the 5 MW light water pool reactor at Aldermaston and its merits for certain kinds of experiment are noted.

The present pattern of operation of the DIDO reactors is then described giving some details of costs, manpower, operating schedules, reactivity, fuel cycles, etc. The importance of well tried and highly reliable experimental equipment is emphasised. Examples of simple and general purpose equipment are quoted.

The scale of design, commissioning and supporting effort for such a programme is described with some reference to the organisational and safety arrangements that have been found most satisfactory.

Many of the physics measurements of flux and reactivity needed to design new reactor equipment and experiments and to plan changes in fuel and absorber loading are now carried out in a zero energy model of the reactors, known as DAPHNE, situated at Harwell. Some typical examples are illustrated.

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Brief mention is made of the Health Physics arrangement to detect and control leakage of radioactive species from either the reactor or experiments into the environment. Experience of tritium is particularly discussed.

The requirements for even higher flux thermal neutron beams have given rise to discussions about new systems devoted explicitly to neutron beam work. A brief review is made of the potential for further development of general purpose reactors of this class.

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Abstract

THE TREND TOWARDS HIGHER SPECIFIC RATINGS IN THE MAGNOX REACTOR IN BRITAIN

by

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J. M. Yellowlees

The specific rating of a reactor directly affects the size of core, the pressure vessel, the biological shield and the site layout. These items are relatively expensive in magnox stations and hence there exists a prima facie incentive to go to higher ratings.

The specific reactor rating depends almost entirely on fuel rating and Vm/Vu.

Physics imposes tight limits on Vm/Vu hence increased specific rating means in practice increased fuel rating.

Fuel rating is limited by:

1. Endurance of fuel element and reactor core
2. Fuel handling (blow-out)
3. Safety

The problems associated with these three items tend to increase as fuel rating

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is increased, but this can be mitigated by appropriate choice of design features. For example, buffetting of the fuel element in the coolant gas stream will increase with fuel rating (higher coolant flow rate) but this can be offset by increasing the coolant pressure. Alternatively a more robust fuel element can be used, or the fuel surface to volume ratio might be increased leading to a lower channel pressure drop characteristic (at the expense of a reactivity penalty).

The British nuclear power stations from Berkeley to Wylfa show a significant rise in mean fuel rating, and associated with this a marked reduction in station construction costs. The increase in thermal ratings has demanded improved heat transfer surfaces for the fuel element and these have, in general, been provided by more closely packed and thinner magnox fins. The improved heat transfer performance has been associated with an increase in the pressure drop across the fuel channels. Gas pressures have therefore been increased in order to avoid excessive blower powers, and also to reduce the gas forces acting on the relatively weak fins. The introduction of prestressed concrete pressure vessels for the Oldbury and Wylfa designs has permitted higher gas pressures to be provided economically.

With solid uranium metal rods the mean fuel rating cannot be increased economically above about 1 MW(e)/Te. For higher ratings it is necessary to employ alternative fuel geometries having thinner sections and an enhanced surface heat transfer capability; for example a cluster of solid rods, tubular element, or a rectangular bar. In an economic comparison of these geometries the important characteristics are fuel fabrication cost and maximum permissible fuel rating. The former is influenced by the complexity of the assembly and the weight of uranium per assembly. The limiting fuel rating may be set by temperature limits, available reactivity, or heat transfer performance. The paper discusses the technical problems and economic trends of alternative fuel geometries within these limitations.

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Abstract

THE ROLE OF WINDSCALE ADVANCED GAS COOLED REACTOR IN THE DEVELOPMENT OF A COMMERCIAL POWER SYSTEM

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\+ T. Marsham
* J. Moore

Operating experience obtained during the first year's operation of the Windscale Advanced Gas Cooled Reactor (WAGR) is described and the role of the reactor and its supporting research facilities in the development programme are highlighted in a programme chart. The design and commissioning of the reactor are not described and readers interested in them are referred to the proceedings of the Symposium on the Advanced Gas Cooled Reactor sponsored by the British Nuclear Engineering Society and held in March 1963.

Operational experience with the WAGR is reviewed in the light of its application to the design and operation of a commercial reactor with specific reference to:-

(a) high integrity of the plant leading to high utilisation and the ability to achieve substantial increases in output over design figures
(b) satisfactory experience with the instrumentation and control systems
(c) behaviour of plant components, in particular those of a novel nature
(d) experience with installed safety circuits.

* The Reactor Group, U.K.A.E.A.
\+ The Production Group, U.K.A.E.A.

A section on the reactor core discusses the problem of planning reactor loading patterns and fuel cycles, and experience with the computer programmes used for the WAGR and presently being developed for commercial AGR cores.

The utilisation of the reactor and the independent loop circuits for the following experimental purposes is described.

(a) As a fuel test bed to test a number of fuel design features in addition to irradiation of fuel representative of commercial AGR designs. The loop circuits are used for experiments of a more advanced nature for example the behaviour of a fuel element containing a simulated burst was studied in a loop prior to its transfer to the main core.

(b) To monitor the graphite core for core movements and for dimensional changes in diameter and length. The reactor and the loops have been used for experiments with different gas mixtures to determine the optimum coolant gas composition and to obtain data on the effects of irradiation on the physical properties of graphite.

(c) Short transient, oscillator, noise and fault simulation experiments have provided data on temperature coefficients and behaviour of the auto control system and permitted predictions relevant to major fault conditions in the WAGR and commercial AGRs. Long term reactivity changes are measured as burn-up proceeds and compared with the results from calculations.

(d) Basic methods for physics calculations are tested against data derived from experiments on sub-critical facilities, from the WAGR during operation and from post-irradiation examination of fuel.

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Abstract

INSTRUMENTATION AND CONTROL OF LARGE GAS-COOLED REACTORS

by

* D. S. Hiorns

+ T. O. Jeffries
+ D. Moore
+ H. Moores
* A. H. Weaving

Since the design of the first British gas-cooled power reactors some 5 - 7 years ago, work has proceeded to simplify operational control and improve reliability. In this paper three aspects of this are discussed, improvements to main control displays, and the application of solid state techniques to safety circuits and to control rod supplies.

The application of large data processing units, now available in a proved and reliable form has facilitated increased centralisation of the control function. The information from the plant amounting to perhaps 7,000 readings on a two reactor station is processed and only the information required by the operators is displayed. The use of cathode ray tubes on which alpha-numeric information can be reproduced, provides the necessary flexibility for this application. Corresponding improvements are also described for the panels housing the controls.

Until recently, all the British civil nuclear power reactors have employed electro magnetic relays in the protection systems. To improve the inherent safety of a protection system, and reduce maintenance testing to a minimum, the application of solid state switching devices has been studied and the "Laddic" has been found to fulfil safety requirements. The "Laddic" is a multi-aperture

* Central Electricity Generating Board, Design and Construction Department, London

+ English Electric Company Limited, Whetstone, Nr. Leicester.

ferrite core, upon which logic may be performed and its development for this purpose is described together with its application to a reactor protection system.

Low frequency supplies in the range 0-1 cycle/second, are provided to operate the control rod actuator in power reactors. This arrangement obviates the need for clutches and high ratio gear trains within the reactor. The low frequency supplies, derived in the early reactors from rotating plant, are now derived from static generating equipment employing magnetic amplifiers or thyristors, which modulate the output of the equipment to give the low frequency required. To guard against the lack of inertia in a static generating system features are incorporated to maintain the low frequency supplies during transient disturbances on the mains supply, thus enhancing reactor availability.

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Abstract

DEVELOPMENTS OF COMMISSIONING PROCEDURES FOR GAS COOLED REACTORS

by

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* E. Stephenson

The planned and actual commissioning programmes for Bradwell and Latina are compared and it is shown that the experience with the earlier stations enabled substantial reductions to be made in the time taken to commission Latina. In the mechanical proving tests, the major savings were in leak testing and rectification, in the testing of the fuel handling equipment, in the removal of protective coatings and in the final proving runs.

A considerable part of the commissioning phase is occupied in loading fuel and in setting the gags controlling the flow of coolant gas in each fuel channel or in measuring this flow.

Improved methods and organisation enabled the times for these operations to be substantially reduced and an outline of these methods is included in the paper.

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* The Nuclear Power Group, Knutsford.

The scope of the physics commissioning programme, prior to power operation, is discussed in relation to the three main objectives (a) setting up the reactor for optimum operation (b) assessment of reactor safety and (c) obtaining information for operational purposes. The complementary roles of experiment and design calculations in achieving these objectives are described and it is shown that confidence in the accuracy of the latter methods is of primary importance.

In order to demonstrate the degree of confidence achieved, the experimental results of the Latina programme are compared with predictions made available before the experiments commenced and it is shown that in general good agreement was obtained. The reduced experimental programme at Latina is explained by comparison with the corresponding Bradwell programme and it is shown by reference to the proposed Dungeness programme that still further reductions can be anticipated in most phases of the commissioning period. In particular, the time now allocated to physics measurements is no more than two weeks.

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Abstract

EARLY OPERATION OF THE C.E.G.B. NUCLEAR POWER STATIONS

by

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*R.L.Rutter

The paper states the objectives before the C.E.G.B. operators when generation commenced in June 1962. It records in tabular form the performance of Berkeley and Bradwell during the peak loading of the C.E.G.B. system in the first winter of operation, November 1962/February 1963. An analysis is made of the station utilisation from start-up to December 1963, and the limitations of output and availability are recorded and discussed. The reliability of nuclear generation following a careful period of commissioning is made clear.

A section deals with the operating costs, records the fuel, fuel handling and other operational and maintenance costs for both Berkeley and Bradwell and makes clear the very high incentive to maintain the nuclear stations at full power for the maximum time. The replacement fuel costs for Bradwell within the C.E.G.B. system for different times of the year are reported. There is discussion of the overall initial performance and comparisons are made between nuclear and non-nuclear effects.

Two of the nuclear limitations to performance are discussed in some detail; the case for a higher permissible fuel element temperature than that adopted at start-up and the case for a higher permitted graphite temperature. The case for the higher permissible fuel element temperature discusses the magnox ignition temperature and records the acceptance of 645°C for this purpose in dry carbon dioxide. The results of reassessments made of the transient rise

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* U.K. Central Electricity Generating Board.

in temperature in the reactor during certain postulated fault conditions, are presented. The case for the higher permissible graphite temperature involved consideration of the chemical reactivity of the graphite from which the graphite cores at Berkeley and Bradwell were constructed and an analysis of the results of sample measurements. The approach to the interpretation of air reactivity of graphite therefore emphasised two points. The first was that the value of graphite air reactivity should properly be associated with a given volume and probability, and the second was that more experimental information on the distribution of reactivity within graphite bricks was needed. A summary is made of the main arguments leading to the acceptance of both cases by the C.E.G.B. Nuclear Power Station Safety Committee and by the U.K. Inspectorate of Nuclear Installations.

The paper also reviews certain operational procedures and discusses the internal inspection of reactors and the case for the internal inspection of the nuclear boilers. New techniques for the former are described and new judgments concerning the latter are suggested.

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Abstract

GENERAL DESIGN OF THE STEAM GENERATING HEAVY WATER REACTOR

by

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\+ A. Firth

= J. E. R. Holmes

The S.G.H.W.R. evolved from the steam cooled heavy-water reactor which in turn evolved from the gas cooled heavy-water studies. The combination of design features in the S.G.H.W.R. results in low capital costs which is mainly a result of low heavy water inventory and a direct steam cycle. Fuel costs are also low due to good neutron economy. On-load refuelling and superheat is possible and large size units can be built without introducing new feasibility problems.

A 100 MW(e) S.G.H.W.R. prototype was selected for the U. K. development programme after separate surveys of light and heavy water reactors by the A.E.A. and by the nuclear Consortia. Development of nuclear superheat is provided for.

The fuel rods are of full core length and cluster spacing is obtained either by grids or by helical wire wrapping. In the boiling channels the fuel is clad in Zircaloy 2: the superheat fuel is clad in stainless steel.

The core comprises 112 Zircaloy pressure tubes (including 8 superheat channels) passing through an aluminium alloy calandria. Light water recirculates through the boiling channels, giving 13% exit steam quality by weight. Stainless steel was selected for the primary circuit with type 410 transition pieces to the pressure tubes.

Reactor power is controlled by moderator level, with moderator poison for long term reactivity changes and absorber fluid tubes for rapid shut down and possibly for flux trimming.

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* U. K. Atomic Energy Authority, Reactor Group, Risley

\+ U. K. Atomic Energy Authority, Reactor Group, Risley

= U. K. Atomic Energy Authority, A. E. E., Winfrith Heath

Some Zr/Nb alloy pressure tubes may be included. Both flanged and expanded pressure tube joints between Zircaloy and stainless iron have been successfully developed. Expanded joints are preferred for their simplicity and replacement possibilities. The superheat channels have similar pressure tubes which are cooled and protected from superheat temperatures by a water cooled annulus and internal liner tube.

The heavy water in the moderator passes downwards through the core after cooling the calandria upper regions. Light water shield tanks surround the calandria.

The core assembly is housed in a concrete shielding vault which also forms a primary containment. A primary circuit failure would first be released to atmosphere through a condensing lute but before fission products could emerge the discharge would be diverted to the reactor hall, which surrounds the primary containment and is equipped with chemical clean up plant.

On-load refuelling is provided and the discharged fuel stringers are stored in a pond adjacent to the reactor.

S.G.H.W. physics problems arise from dual moderation and from light water boiling. The reactor is designed to give void coefficient which will ensure stability and safety. Isotopic changes during burn-up must be taken into account. Floodable displacer tubes permit adjustment of void coefficient during burn-up if necessary. Fuel management studies are in progress to determine the most economical method of power flattening.

A number of experimental features include loops for special irradiation and chemical experiments, and extra instrumentation.

The reactor is sited at the Winfrith Heath establishment of the Authority.

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Abstract

PERFORMANCE AND OPERATION OF THE DOUNREAY FAST REACTOR

by

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D. M. Donaldson
K. J. Henry
J. L. Phillips
H. E. Tilbe

1. The major engineering difficulties encountered during the commissioning stages of the reactor have been successfully overcome. The original metallic fuel concept has been retained and following redesign of the element as a result of experience on irradiation behaviour, the limiting burn-up of the fuel has been raised to over 1% of heavy atoms. The redesigned element permitted the reactor power to be raised to the design figure of 60 MW. The heat rejection plant has performed very satisfactorily, electrical power has been produced and exported to the National Grid.

2. Having demonstrated the feasibility of operating fast reactors at power, the reactor has now been converted into a valuable fast flux irradiation test facility capable of irradiating sub-assemblies, experimental fuel pins of varying lengths and specimens of canning materials.

3. Modifications to the reactor circuits have been made to increase the capacity of the cold trapping system and new in-line purity measuring devices have been successfully developed. Other additions have been a corrosion sampling device which is operable at full circuit pressure and a corrosion measuring device in one of the by-pass circuits. Certain other modifications have been made as required by experience with activity in the cover gas arising from the use of vented fuel elements.

All authors U.K. Atomic Energy Authority, Reactor Group, Dounreay

4. A description is given of the major steps in starting up and shutting down the reactor, together with an outline of the method of control.

5. Performance of plant components has been generally good, particularly the instrumentation and safety circuits. No unscheduled shut downs from power have been necessary, nor initiated by instrument failures.

6. The kinetic behaviour of the reactor has been extensively studied and generally is in agreement with prediction. The stability of the reactor has been established throughout the entire operating range, and results indicate that prompt positive coefficients have been successfully eliminated by the design adopted.

7. With the achievement of full power operation, experience has been rapidly gained on the control of the fission product activity both during normal operation and during maintenance operations. The chemistry of the coolant has been studied in so far as compatibility of the canning materials is concerned, and the results of reactor and loop experiments are compared.

8. Fabrication of the redesigned metallic fuel element has not given rise to any particular difficulty. Operational experience with the fuel element charge machines and improvements giving increased rates of handling elements are described.

9. The chemical reprocessing plant has now commenced operation with highly irradiated fuel and the mechanical removal of the can has given rise to several problems arising from the changed nature of the fuel and canning materials. Very satisfactory techniques and equipment have been evolved for the dissolution process and the fission product decontamination in the separation plant has been very efficient. The problem of solvent degradation has not been so severe as expected mainly because of the efficiency of the first wash process.

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Abstract

THE INITIATION OF A NATIONAL NUCLEAR POWER PROGRAMME

by

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R.F.W. Guard +

This Paper discusses some of the practical problems facing a country about to instal its first nuclear power reactor. It is assumed that the country concerned is one which normally imports heavy plant, and will expect to make its first power reactor the subject of commercial contracts with foreign suppliers. A great deal of preparatory work needs to be carried out before such contracts can be signed, and it is the object of the present Paper to survey this work and point out some of the difficulties that arise.

For a number of reasons a country may decide to build a nuclear power station whether or not it is clearly the most economic form of generation. Once built, however, the station must be run in such a way that the overall operating cost of the entire interconnected electricity system is minimised. Therefore a long-term study of the national power requirements must be made before the optimum size of station can be calculated.

It will be recognised that out of all possible reactor systems only very few can be regarded as "proved". The Paper summarises the practical experience of design and operation of the major reactor types, and discusses the potentialities of some of the systems still only in the prototype stage. Comments are made on the principal areas of uncertainty.

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The choice of site follows normal thermal generation practice with the addition of certain special requirements. National regulations for population exclusion, the emission and disposal of radioactive materials, third party nuclear insurance and other legislative matters, require drawing up after study of national and international experience elsewhere.

Once the basic decisions to proceed have been taken, an administrative organisation must be set up to manage the project, reinforced as required by consultants and specialists. This group must set themselves target dates for completing the preliminary work which is the purchaser's responsibility, followed by the placing of the main construction contracts and the achievement of full power generation by a certain date.

The ideal Invitation to Tender includes a clear Specification which ensures that the bids received may be properly evaluated and compared. The purchaser should propose Conditions of Contract which will lead to the minimum delay in negotiating the final contracts. A good contract is satisfactory to both parties, and in the nuclear field the purchaser should understand what a reactor designer can and cannot be reasonably expected to guarantee in the present state of the art.

The comparison of a number of tenders for widely different reactor systems raises some interesting problems. It is desirable to establish as early as possible the economic ground rules to be used in making an economic comparison, which may not lead to the choice of the lowest capital cost system. Energy cost may not even be the principal criterion, as the calculations may be "weighted" by considerations peculiar to the purchaser, such as the degree of local industrial participation desired. These considerations should be mentioned in the Invitation.

The coordination and supervision of work on site follow the placing of the principal contracts.

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Abstract

FISSION HEATED THERMIONIC DIODES

by

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The scope and objects of the United Kingdom Atomic Energy Authority programme on fission heated thermionic converters are described. Two types of inpile rigs were operated in 1962/63, one with a fixed electrical load and one with a variable electrical load. All these rigs (four) had unclad UC-ZrC rods as emitters and their output current was of the order 10-15 amps cm^{-2} of the emitter surface. Post irradiation examination indicates some fuel material was evaporated from the emitter and was deposited on the anode. There are indications that UC evaporates preferentially from the UC-ZrC alloy.

The present development programme is outlined. To obtain a more accurate measure of the efficiency of the diodes future rigs will be placed inside calorimeters. Rigs having refractory metal emitters operating at elevated caesium pressures have been constructed and a rig consisting of four generators in series inside a single envelope has been designed.

Fabrication of UC-ZrC fuel, compatibility tests performed on various materials in caesium and plans for future work on refractory metal-fissile fuel compatibility and ceramic to metal seals are discussed.

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Abstract

*NUCLEAR MAGNETOPLASMADYNAMIC ENERGY CONVERSION

by
+B. C. Lindley

Research on closed-cycle magnetoplasmadynamic (MPD) generator systems, employing alkali metal seeded inert gases as the working fluid, is at an early stage. The nuclear heat source is the most difficult problem, but a direct gas-cooled reactor may be technically feasible. For thermal ionization, the electrical conductivities of cesium-seeded inert gases are comparable with those of potassium-seeded combustion gases at temperatures about 500°C higher, so that the closed-cycle has an immediate advantage over the coal- or oil-fired open cycle. There also exists the possibility in inert gas plasmas of achieving high electrical conductivities through non-equilibrium ionization at quite moderate gas temperatures (1200°-1800°C). If operation at such low temperatures is proved, the nuclear closed-cycle scheme becomes much more attractive.

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* The research programme at IRD on MPD power generation is sponsored jointly by C. A. Parsons and Company Limited and the Advanced Research Projects Agency of the US Department of Defense under US Navy Contract No. N62558-3127.

+ International Research & Development Co. Ltd., Newcastle upon Tyne, 6, England

An attempt is made to evaluate the closed-cycle MPD concept in terms of technical and economic feasibility, and the associated long-term implications on electrical power generation. A broad research programme, based on a pilot-scale closed-loop helium-cesium facility and a series of supporting plasma physics experiments, has been undertaken at the IRD laboratories. Constructional techniques for circuits operating at gas temperatures up to 2250°C and progress in evaluating the physical processes of MPD generation are described. An assessment of the critical operating parameters, such as overall cycle efficiency and generator power density, is included.

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Abstract

EXPERIMENTAL EQUIPMENT ASSOCIATED WITH THE U.K. HIGH-FLUX RESEARCH REACTORS

By

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DIDO, PLUTO and DMTR are general purpose high-flux research reactors, which have been in operation for many years and which are now intensively used. The diversified use of these reactors has led to a wide variety of designs of experimental equipment and the most interesting and important of these have been chosen for discussion.

A substantial part of the irradiation programme has been concerned with the behaviour of non-fissile specimens in high fast neutron fluxes. The evolution of rigs for this type of experiment is described. The original design was for irradiating graphite samples at a few hundred degrees Centigrade, while later rigs now accommodate a wide variety of materials at temperatures up to 1500°C.

A rig for fissile specimens which has similarly undergone several stages of development is also described.

The high-pressure water loop in DIDO has been working satisfactorily for several years. The principal features of the loop are outlined and modifications to allow up to four individual experimental holes to be cooled by parallel water currents from the one high-pressure water system are discussed.

The High-Temperature Gas-Cooled Loop, which is installed in PLUTO has been used exclusively by the O.E.C.D. DRAGON Project. The design concept of this loop, together with some commissioning problems and operational experience using possible DRAGON Reactor fuel elements is recounted.

Quite different problems have arisen in the provision of rigs for producing 'cold' (15 - 45°K) and 'hot' ($\sim$1100°K) neutron beams. These are discussed together with the operation of the liquid-hydrogen moderator chamber in DIDO.

Finally, the future requirements for rigs, especially with regard to temperature, are discussed and some of the proposals for meeting them described.

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Abstract

HEAT TRANSFER AND PRESSURE DROP PERFORMANCE OF HERRINGBONE AND HELICAL FIN FUEL ELEMENTS FOR URANIUM/MAGNOX REACTORS

by

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\+ B.E.Boyce
= T.Davis
Ƶ B.N.Furber

Polyzonal fuel elements designed for gas-cooled Uranium-Magnox reactors in Great Britain have either multistart helical or "herringbone" fins as extended surfaces. This paper deals principally with the experimental determination of the heat transfer and pressure drop characteristics of such elements and is divided into four main parts.

The performance data which is necessary for the selection of a suitable fuel element design is discussed in the first section. Test results are presented to illustrate the effects of changes in the salient element parameters, e.g. number or lead of fins, on heat transfer and pressure drop. These results are compared with predictions using previously published correlations.

When a design of fuel element has been selected for a reactor, its performance is required in great detail and to a high accuracy to enable the reactor designer to calculate station output and fuel element temperatures under normal operating conditions. The second part of the paper describes the refinements in test apparatus and a typical experiment design evolved to measure

* Atomic Power Projects, Whetstone, near Leicester
\+ United Power Company, London
= U.K.A.E.A., Reactor Group, Windscale
Ƶ The Nuclear Power Group, Knutsford

heat transfer for these conditions.

The chosen fuel element must also be designed for safe operation when the reactor is running on part load at low coolant flows or when system faults cause partial or virtually complete loss of coolant. Experiments to provide performance data for element assessment under such conditions are described in the third section together with typical results.

The final section deals with two important aspects of polyzonal fuel element performance on which little or no data has been published. These are (a) the influence of random orientation of successive elements in a reactor channel on heat transfer and pressure drop, and (b) the changes in performance associated with deformation of the finning by gas forces. Experimental results are presented which illustrate the magnitude of effects of these two features on heat transfer and pressure drop.

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Abstract

HEAT TRANSFER PERFORMANCE OF THE POLYZONAL CAN FOR THE C.E.G.B. REACTORS

by

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* E.W.V. Acton
\+ J.A. Hitchcock
\+ S.B.H.C. Neal

The paper describes the heat transfer research undertaken by the Central Electricity Generating Board into the polyzonal fuel element. It outlines the problems which are of interest to the reactor operator and which make it necessary to obtain as accurate a correlation as is possible between laboratory data and the performance of the fuel element within the reactor.

The complex nature of both the fluid flow and the associated heat transfer processes around the polyzonal spiral surface which have previously eluded purely theoretical treatment have been investigated using a four times full scale model. The basic understanding which has been gained of these mechanisms is described in the paper and suggestions are made for improving the heat transfer performance of the surface.

Investigations into polyzonal heat transfer performance in the Laboratories have shown that results are critically dependent on the circumferential variation in thickness of the heater tube wall, accumulated effects of manufacturing tolerances and the presence of and errors in thermocouples. Other factors which have been investigated experimentally include the effects of flux peaking at the ends of the fuel elements and the differences between the end of a fuel element in a reactor and the electrically heated laboratory assembly.

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*+ Central Electricity Generating Board, Research & Development Department,
* Berkeley Nuclear Laboratories
\+ Central Electricity Research Laboratories

Measurements of the fuel element heat transfer parameters in a Berkeley Reactor are compared with laboratory results. The deterioration in performance which results from the creep deformation of fins, splitters and braces is discussed and an indication of magnitude of the effects is given. The case of a failed fuel element containing an oxide mound on the surface of the uranium bar has been considered and the thermal performance evaluated from an experiment in which the geometry is simulated and a comparison is made with the results from an analogue study.

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Abstract

GRAPHITE CORE STRUCTURES FOR LARGE GAS-COOLED POWER REACTORS

by

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M. A. Bayer[+]

The core of a gas-cooled graphite moderated reactor consists of columns of graphite pierced by vertical channels for control and for fuel. The structure must be capable of accepting irradiation distortion and differential thermal movements and at the same time be stable against loads due to gas pressure differences, fuel handling, and in some cases seismic forces. For large cores, it is advisable to design the structure so that each column is located independently of its lateral strain and is held vertical by some suitable skeleton. In this way not only is it insensitive to the exact value of irradiation strains but the misalignment of columns is kept to a minimum. The usual solution is to use radial keys on the graphite components so as to form a layer which changes its pitch in a consistent pattern when the boundary is expanded or contracted as a whole. The periphery may be fixed to a steel structure, or to a graphite arch held in place by elastic garters compensated for temperature. Other structural problems include bowing of blocks caused by differential shrinkage and gas bursting forces especially under burst duct fault conditions. The other main problem in core design is to prevent gas by-passing fuel channels. The spaces between bricks have to be vented to the top of the core because of control requirements so the bottom layer has to seal across the base either in the form of a steel plate or by butting reflector blocks solid. Gas leaks out of fuel

[+]United Power Co. Ltd., London.

channels through the permeable graphite and through the joints between blocks, and a system without sleeves should incorporate special sealing features between blocks to prevent excessive leakage after long term distortion.

The Hunterston, Trawsfynydd, and Tokai Mura core designs all embody keyed layers to cater for thermal and irradiation strains, but Trawsfynydd expands as graphite, Tokai Mura as steel, and Hunterston uses a mixed solution. A study of these core designs shows how the problems have manifested themselves in a slightly different way on each one and have resulted in different solutions and development requirements. It is interesting to consider how the same problems manifest themselves on a current large magnox unit for 500 MW n.e.o. and to describe a core design for it based on the experience of the three earlier reactors. This leads to a core with deep keyed layers alternating with plain layers, spiggoted into a steel at the bottom and at the periphery. Similar principles can be adopted on a core for an advanced gas reactor.